AUG 0 9 1984

Docket Nos.: 50-445
and 50-446

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: Resolution of SER Confirmatory TMI Action Plan Items and Status of Staff Review on Steam Generator Tube Rupture Event Relative to the Comanche Peak Steam Electric Station (Units 1 and 2)

Enclosed are the staff's evaluation findings relative to the following TMI Action Plan Items (SER Confirmatory Issue 5), resolving those items, which we propose to incorporate in a future SER supplement.

Item

II.B.1
II.K.2.13
II.K.2.17
II.K.3.1
II.K.3.2
II.K.3.5
II.K.3.30
II.K.3.31

Also included in the enclosure is the status of the NRC staff's review of the Steam Generator Tube Rupture event (New Issues Raised by NRC #1 listed in my letter to R. J. Gary dated January 24, 1984). My letter to you dated April 9, 1984 requested additional information needed by the staff to complete its review of the Steam Generator Tube Rupture event which is awaited.

Sincerely,

/s/

B. J. Youngblood, Chief
Licensing Branch No. 1
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Enclosure:
As stated

cc: See next page

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ENCLOSURE 1

II.B.1 Reactor Coolant System Vents

As stated in the SER, this TMI Action Plan item requires the applicant to provide venting capability for Reactor Coolant System high points so that adequate core cooling can be maintained, especially for events beyond the present design basis.

In FSAR Amendment 38 and by a letter from R. A. Werner, Texas Utilities Generating Company, to B. J. Youngblood, NRC, June 28, 1984, the applicant described the high point vents for CPSES. Venting of the reactor vessel head is provided via a line containing a 3/4" orifice and two one-inch valves in series. The line vents directly to the containment atmosphere in an area that provides good mixing. The venting of the pressurizer vapor space is provided in the same manner. The valves are fail-closed solenoid valves which are environmentally and seismically qualified and powered by class IE power supplies. Therefore, a single failure of a vent valve or a power supply will not prevent isolation of the vent paths. Furthermore, if the two vent valves were left open, the flow out of the vent line would be less than that of a small break LOCA and within the capacity of the reactor makeup system.

The vent valves are provided with positive valve position indication in the form of red/open and green/closed indicating lights on the control panel. Also, an audible control room alarm will sound if any vent valve is not fully closed. These valves are operable from the control room and will be tested in accordance with Section XI of the ASME code.

The applicant stated that the CPSES plant will use emergency procedures consistent with the Emergency Response Guidelines (ERGs) developed by the Westinghouse Owners Group and approved by the staff by a memorandum from R. Mattson, Director, DSI, and H. Thompson, Director, DHFS, to D. Eisenhut, Director, DL, May 18, 1983. The staff evaluation of the ERGs concludes that high point venting guidelines are addressed adequately and are, therefore, acceptable.

The staff finds that the applicant has satisfied the requirements set forth in the TMI Action Plan Item II.B.1.

II.K.2.13 <u>Thermal Mechanical Report</u>

As stated in the SER, the Westinghouse Owners Group (of which the applicant is a member) submittal regarding the thermal mechanical analysis was being reviewed by the staff. The staff has recently completed its review and concluded that the information provided is adequate in demonstrating reasonable assurance that vessel integrity is maintained for a loss-of-feedwater induced LOCA aggravated by loss of auxiliary feedwater. The staff's conclusions regarding this issue are based on findings related to USI A-49 "Pressurized Thermal Shock." The evaluation of industry responses to TMI Action Item II.K.2.13 is contained in the NRC letter to PWR licensees: "Safety Evaluation By The Office of Nuclear Reactor Regulation Concerning NUREG-0737 Item II.K.2.13, Thermal Mechanical Report, Effect of High Pressure Injection on Vessel Integrity for Small Break Loss-of-Coolant Accident with no Auxiliary Feedwater For All Operating Pressurized Water Reactor Plants," June 5, 1984. Based on its review, the staff finds that the applicant has satisfied the requirements set forth in TMI Action Item II.K.2.13

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

As stated in the SER, the TMI-2 Action Plan requires the applicant to analyze the potential for voiding in the reactor coolant system during anticipated transients. Westinghouse, in support of the Westinghouse Owners Group (of which CPSES is a member), has performed a study of the potential for void formation in Westinghouse plants during natural circulation cooldown and depressurization transients. The study has been submitted to the NRC by the Westinghouse Owners Group Letter OG-57, R. W. Jurgensen to P. S. Check, April 20, 1981.

The staff's evaluation of this report is contained in the memorandum from R. W. Houston to G. Lainas, "Multi Plant Action Item F-33, Voiding in the Reactor Coolant System During Anticipated Transients," December 6, 1983. The staff concludes that the Westinghouse study is applicable to all Westinghouse plants including the Comanche Peak plant. The staff also concludes that the voids generated in the reactor coolant system during anticipated transients are accounted for in present analysis models. Furthermore, based on transient analyses performed by Westinghouse using these models, the staff concludes that steam voids, if formed, will not result in unacceptable consequences during anticipated transients. The staff finds that the applicant has satisfied the requirements set forth in the TMI Action Plan Item II.K.2.17.

II.K.3.1 Installation and Testing of Automatic PORV Isolation System

As stated in the SER, the TMI-2 Action Plan requires that, if the need is determined, the applicant provide a system which automatically closes the PORV block valve to protect against a small-break loss-of-coolant accident. The need for such a system in conformance to II.K.3.1 will be determined following staff evaluation of the Action Plan Item II.K.3.2, "Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System."

The Westinghouse Owners Group submitted WCAP-9804, "Probabilistic Analysis and Operational Data in Response to Item II.K.3.2 for

Westinghouse NSS Plants," to the NRC on March 13, 1981. The staff has reviewed this generic report and concurs that an automatic PORV isolation system is not needed (F. H. Rowsome, DST, to G. C. Lainas, DL, Safety Evaluation of the Westinghouse Licensees' Responses to TMI Action Item II.K.3.2, July 22, 1983). The staff is currently evaluating the applicability of the above generic report to the Comanche Peak plants. However, we note that the CPSES PORVs and Safety Valves (Copes-Vulcan and Crosby, respectively) are of the same types that are in use in other similar Westinghouse plants addressed by WCAP-9804. TMI Action Item II.K.3.2 is addressed in Section 22.2 of the SER.

Based on the staff evaluation of the generic response to item II.K.3.2 (i.e. WCAP-9804), and subject to confirmation that the generic response is applicable to CPSES, the staff finds that the applicant has satisfied the requirements set forth in the TMI Action Item II.K.3.1.

II.K.3.11 Justification of Use of Certain PORVs

As stated in the SER, this item requires the applicant to demonstrate that the PORVs installed in the plant have failure rates equivalent to or less than the valves for which there is an operating history.

The CPSES plants use Copes-Vulcan PORVs with model No. D-100-160. This type of PORV is widely used in operating plants and, therefore, has a substantial operating history. This operating history is addressed by TMI Action Plan Item II.K.3.2.

Subject to the resolution of II.K.3.2 the staff finds that the applicant has satisfied the requirements set forth in II.K.3.11.

II.K.3.5 Automatic Trip of Reactor Coolant Pumps During LOCA

As stated in the SER, the Westinghouse Owner's Group (of which the applicant is a member) submittal has been under staff review. Although the staff review of this submittal has not been completed, this issue is being actively pursued on a generic basis. Based on the applicant's commitment to resolve this issue within the Westinghouse Owner's Group framework and based on the satisfactory progress of the generic review of the issue, the staff considers this issue to be resolved for CPSES.

II.K.3.30 Small Break LOCA Methods, and
II.K.3.31 Plant Specific Calculations

As stated in the SER, the applicant has committed to respond to the staff concerns. The staff has referenced the Westinghouse Owners Group (of which the applicant is a member) submittal regarding this issue. Although the staff's review of this submittal has not been completed, this issue is being actively pursued on a generic basis. Based on the applicant's commitment to resolve this issue within the Westinghouse Owners Group framework and based on the satisfactory progress of the generic review of the issue, the staff considers this issue resolved for CPSES.

15.4.4 Steam Generator Tube Rupture

In its continuing review of the various accidents and transients, the NRC staff reviewed the accident scenario described by CPSES for postulated Steam Generator Tube Rupture (SGTR) events. The sequence of events described assumes that if offsite power is lost, the turbine bypass valves would close and the steam would discharge to the atmosphere through the steam generator atmospheric relief and/or safety valves. Also, the FSAR analysis assumes that if of site power is available, the turbine bypass valves would discharge steam to the condenser. For either case, the FSAR further assumes that the break flow is terminated within 30 minutes after initiation of the event by operator action to equalize primary and secondary pressures. However, SGTR events experienced at operating reactors indicate that a time period much longer than 30 minutes is required to attain pressure equalization; e.g., the R. E. Ginna plant required 3 hours for SGTR Isolation; the Point Beach Plant, 1.75 hours.

In view of this experience, and to more fully evaluate the SGTR accident scenario, the staff held meetings with Westinghouse and several NTOL applicants with Westinghouse nuclear steam supply systems on February 24, 1984, and July 17, 1984, to discuss their plans for adequately resolving the SGTR concern. The meeting summaries provide a brief outline of the issues discussed among the NRC staff, Westinghouse, and the NTOL applicants. Westinghouse will issue a report summarizing its SGTR analysis, tentatively scheduled for September 1984, with a supplement regarding the effects of overfill in March 1985. By letter dated April 9, 1984, the staff requested the applicant to provide additional information related to this area. The staff will report its findings in a supplement to this SER.